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Nuclear data uncertainty propagation

Part 3: ND Uncertainties for reactors and fuel.

EXTEND summer school, Uppsala University, Sweden, September 2nd, 2016

Summary

- Applications to energy systems (*Part 3*)

I. Methods: Monte Carlo (TMC) *vs.* perturbation (sensitivity)

II. Results with TMC
   1. Criticality-safety benchmarks
   2. PWR Fuel pin keff
   3. Assemblies
   4. Full core
   5. Transient

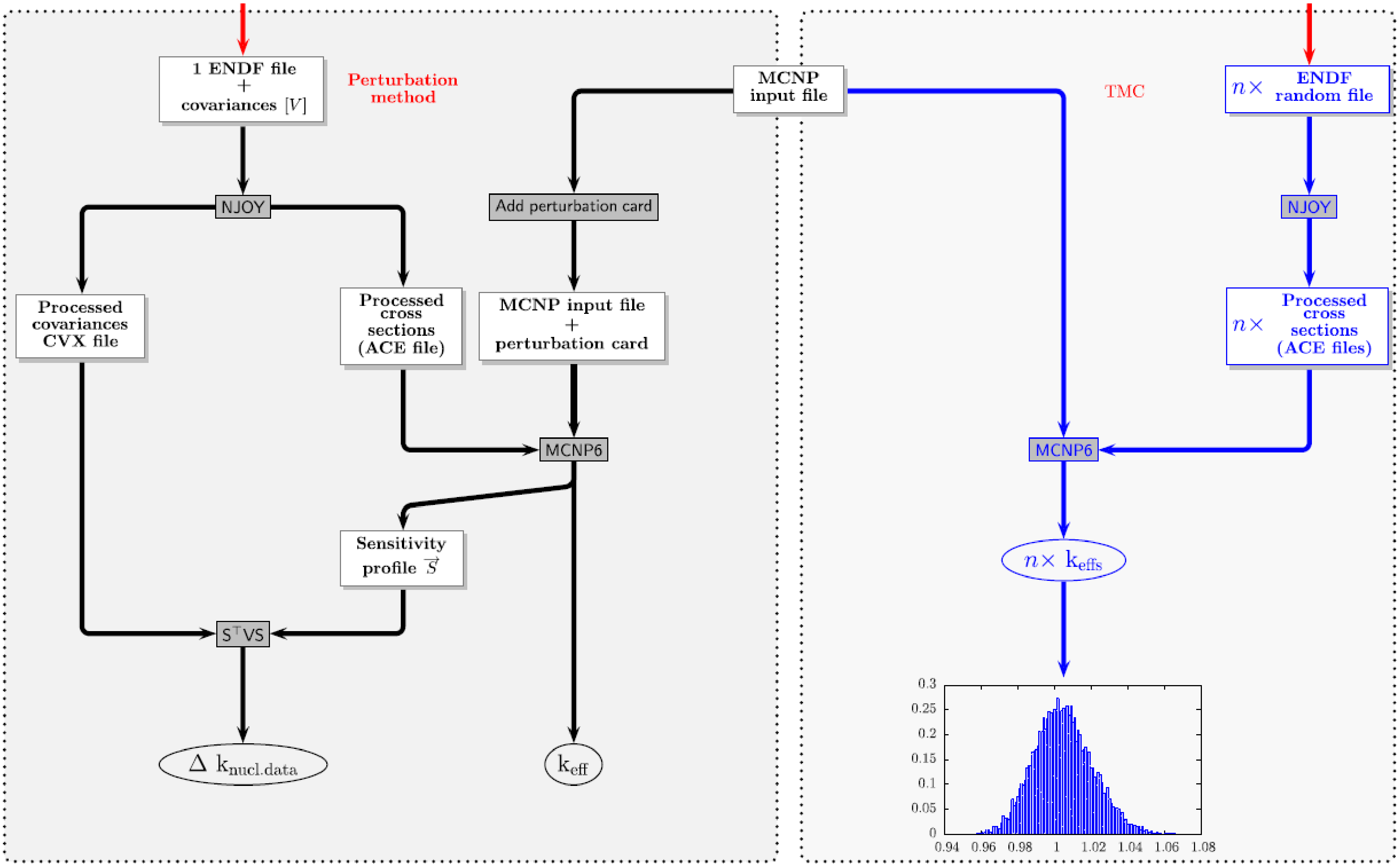
III. Uncertainties from methods

IV. Other uncertainties

All slides can be found here: https://tendl.web.psi.ch/bib_rochman/presentation.html

Methods

- As mentioned before, there are basically two ways of propagating uncertainties



Methods

- In any case, uncertainties are not real quantities, contrary to cross sections

- They are only a reflection of the method applied and of the considered inputs !

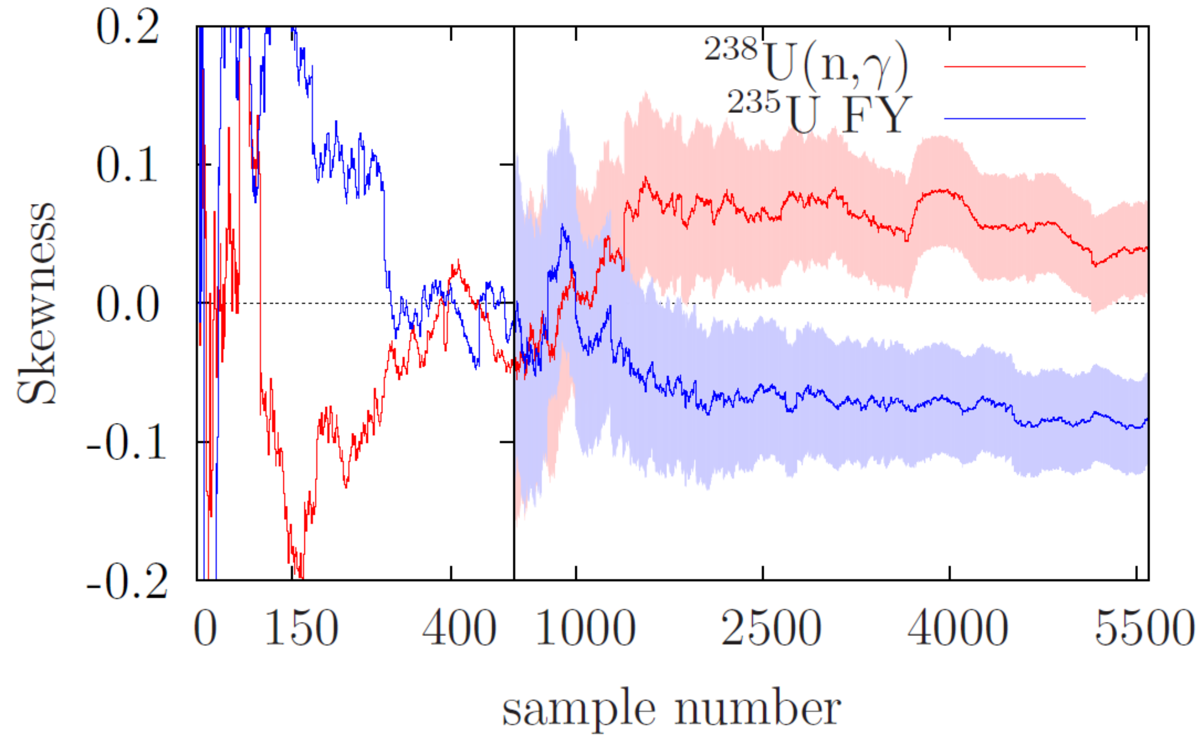
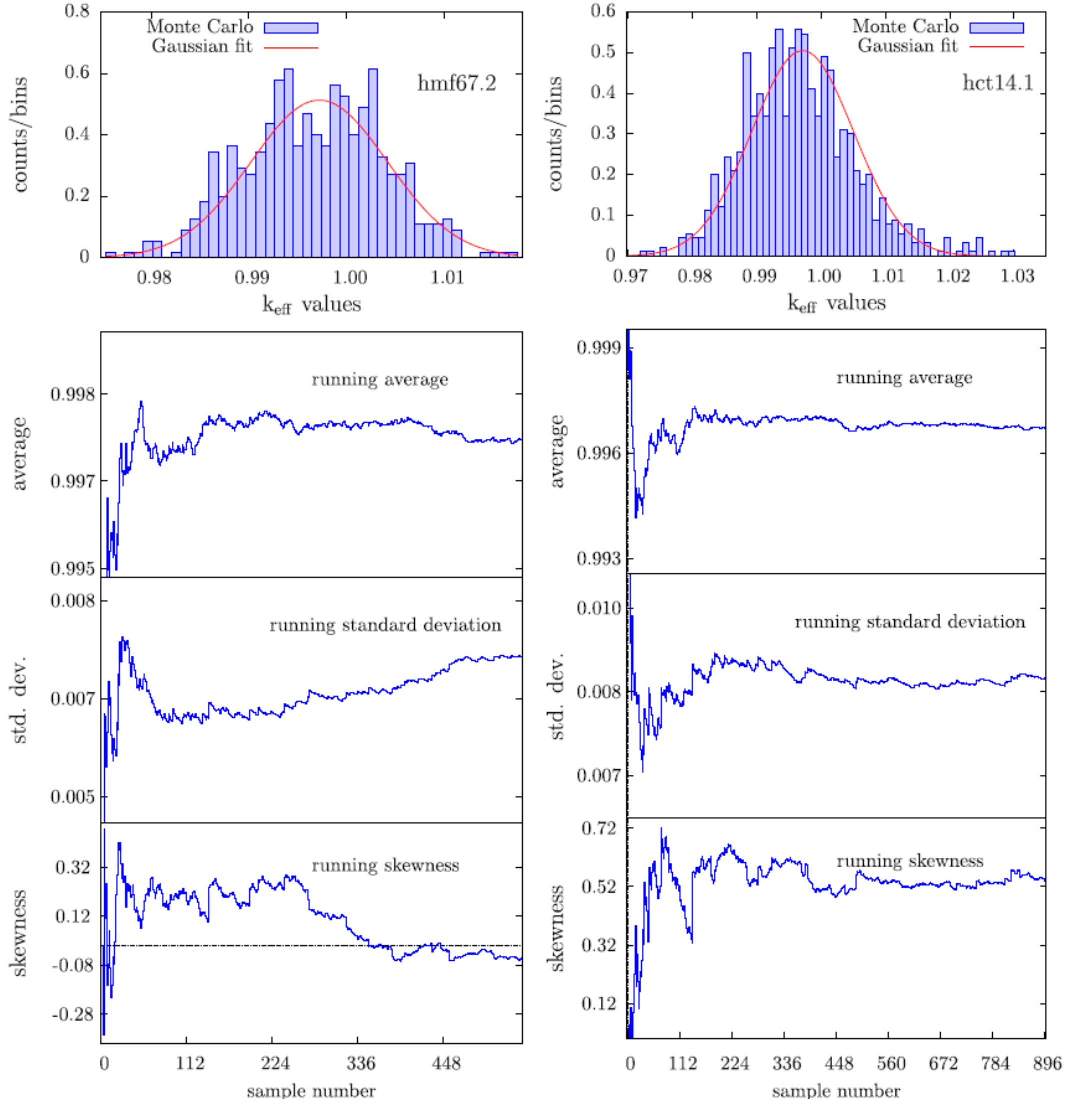


FIG. 46. (Color online) Examples of two different skewness for the same quantity, the number density of the pseudo fission product, changing the ^{238}U(n,γ) cross section at 10 keV in one case (red) and the ^{235}U fission yields in the other case (blue).

Methods

- In Monte Carlo method, the convergence of the results is a key quantity



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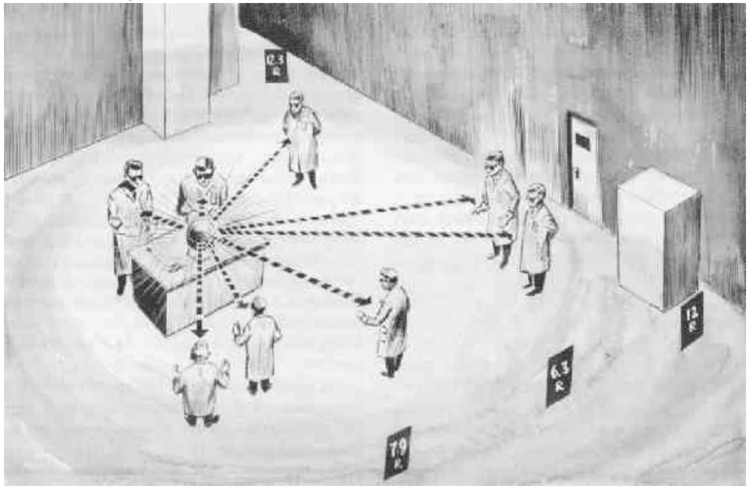
Results 1. Criticality-safety benchmarks

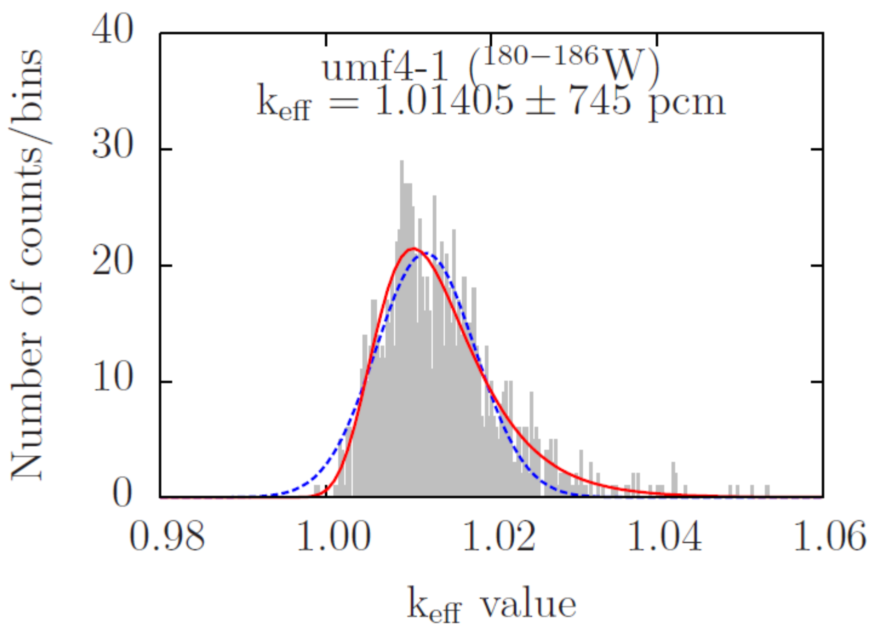
- Criticality-safety benchmarks are crucial to assess the criticality safety of nuclear installation (fuel storage, liquid waste, fissile material storage…)
- It is all about k_{eff} and must be < 1
- Safety authorities often impose an administrative limit of 0.95 (conservative approach)
- Economics pushes for a "best estimate + uncertainties" approach
- As a consequence, 0.95 might not be valid anymore,
- As a consequence, more fissile material can be stored,
- All depending on precise estimation of the uncertainties on k_{eff}.

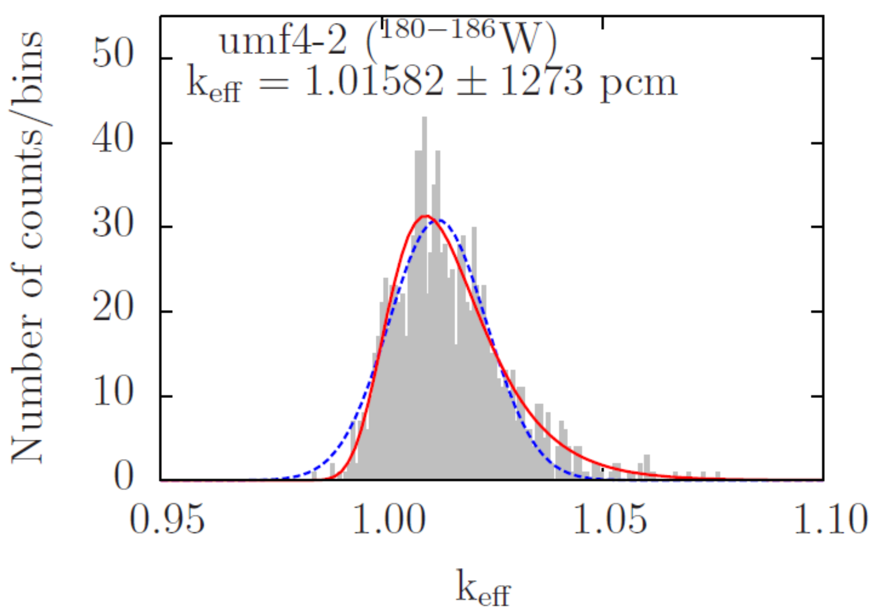


Results 1. Criticality-safety benchmarks

- Criticality benchmarks are used to validate codes (such as MCNP) to make sure that the calculated k_{eff} is correct.

- Additionally, Monte Carlo Uncertainty propagation method lead to “pdf”, more suitable for uncertainty-safety assessments



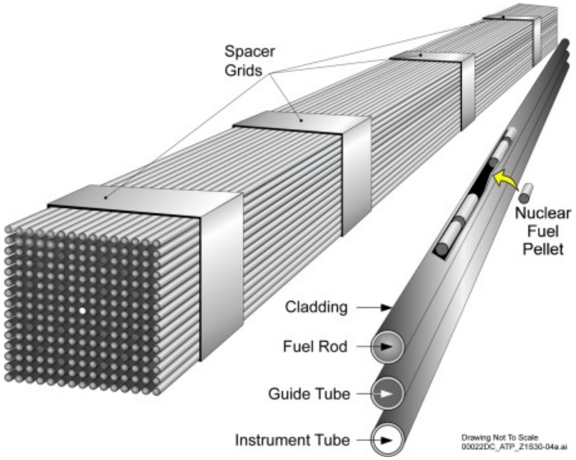


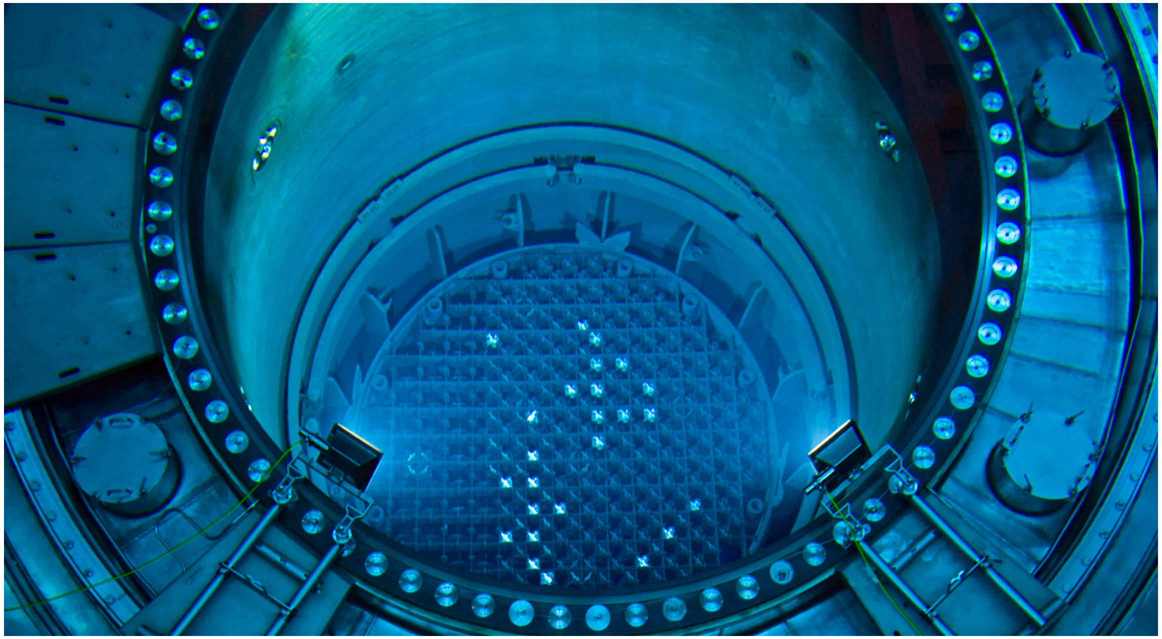
- The strength of the MC methods is in the access to the moments of the distributions, not reflected by a single number such as the standard deviation.

Results 2. PWR Fuel pin

All starts with a pincell:

- Assembly simulations start with pincell simulations,
- Core simulations start with assembly simulations,
- Fuel storage simulations start assembly simulations,





Results 2. PWR Fuel pin

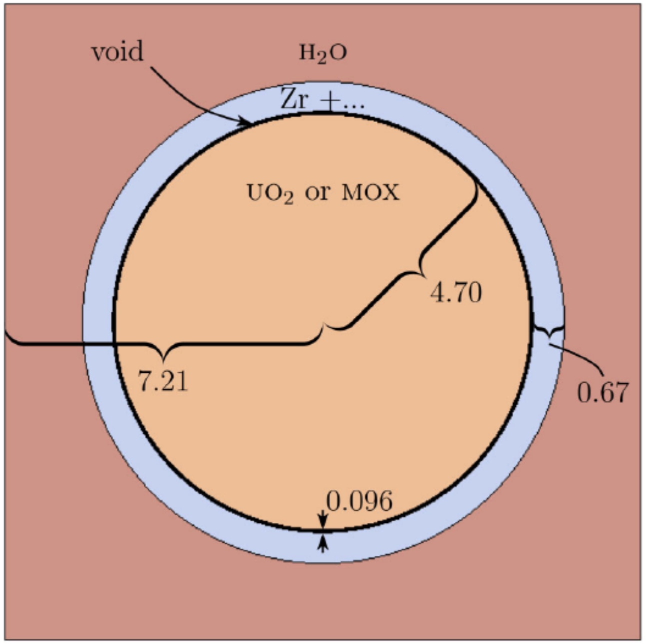


Fig. 1. The geometry of the pin cell model used in Serpent. The fuel, either UO_2 or MOX, is surrounded by concentric annular rings with a void and Zircaloy clad. The rest of the square is filled with water, and all sides are subject to reflecting boundary conditions. All distances are in millimeters.

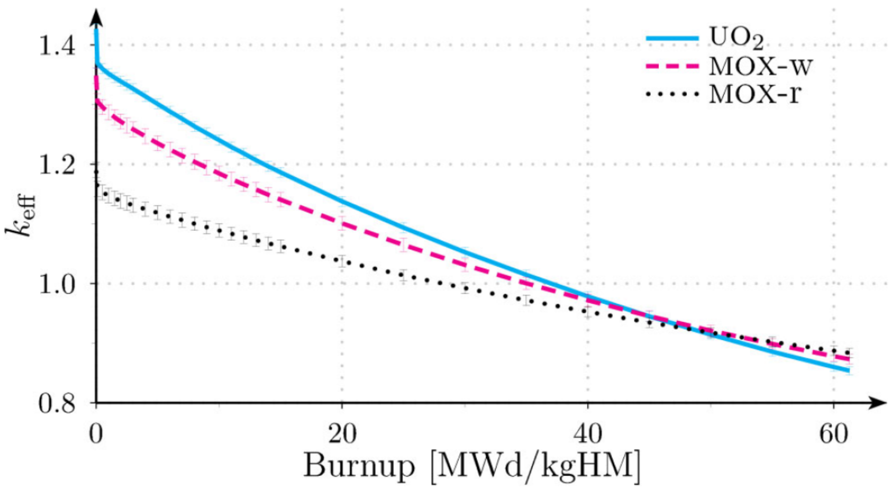


Fig. 3. $k_{\mathrm{eff}} = k_\infty$ as a function of burnup for the three fuel types. The large deviations from 1 are explained by the simplified model: no leakage, infinite grid of pin cells (with the same burnup), and no control mechanisms. The uncertainty bars represent the data uncertainty $\sigma_{data}(k_{eff})$; the statistical uncertainty is negligible in comparison.

Results 2. PWR Fuel pin

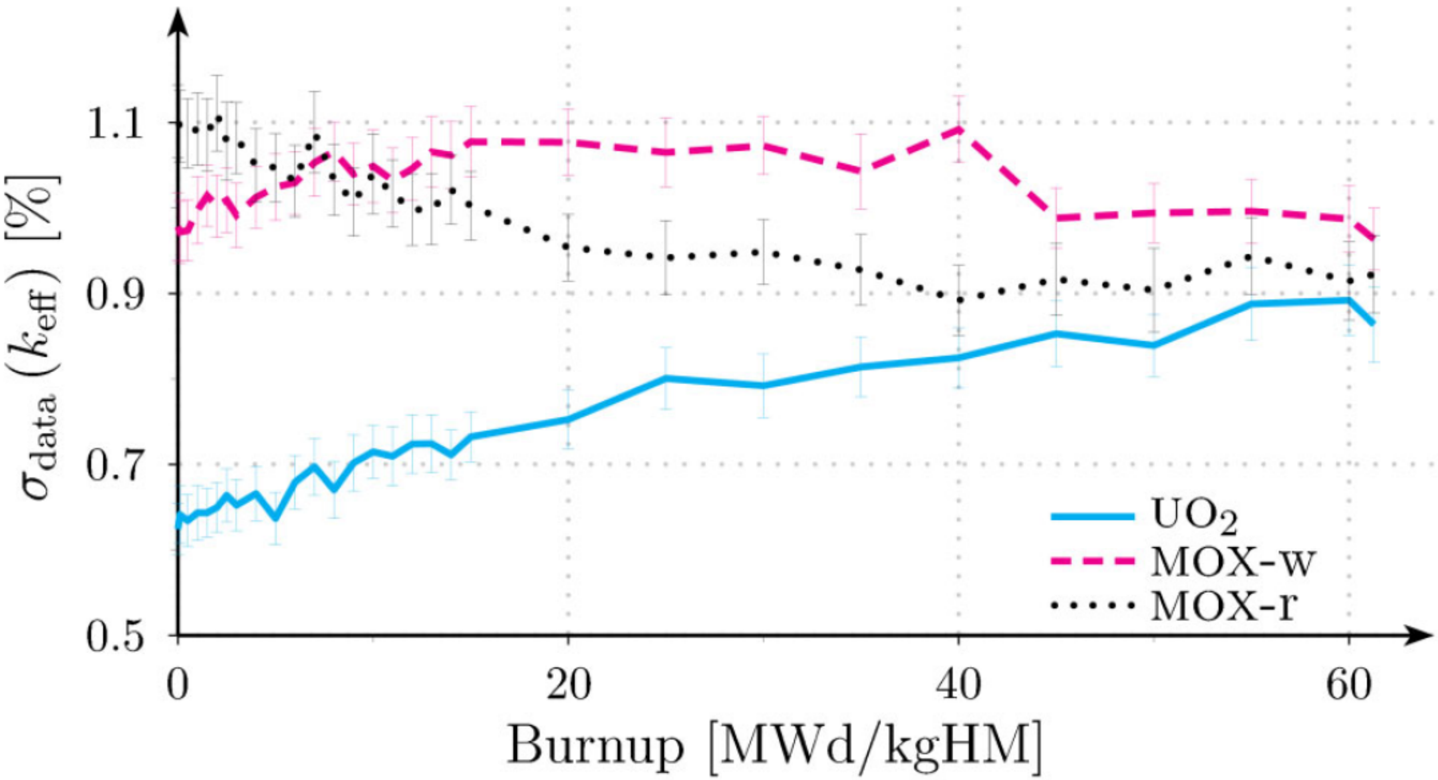


Fig. 2. The main result: Propagated data uncertainty in k_{eff} for UO_2 and the two types of MOX fuel as functions of burnup due to all data. The uncertainty bars represent one standard deviation.

Results 2. PWR Fuel pin

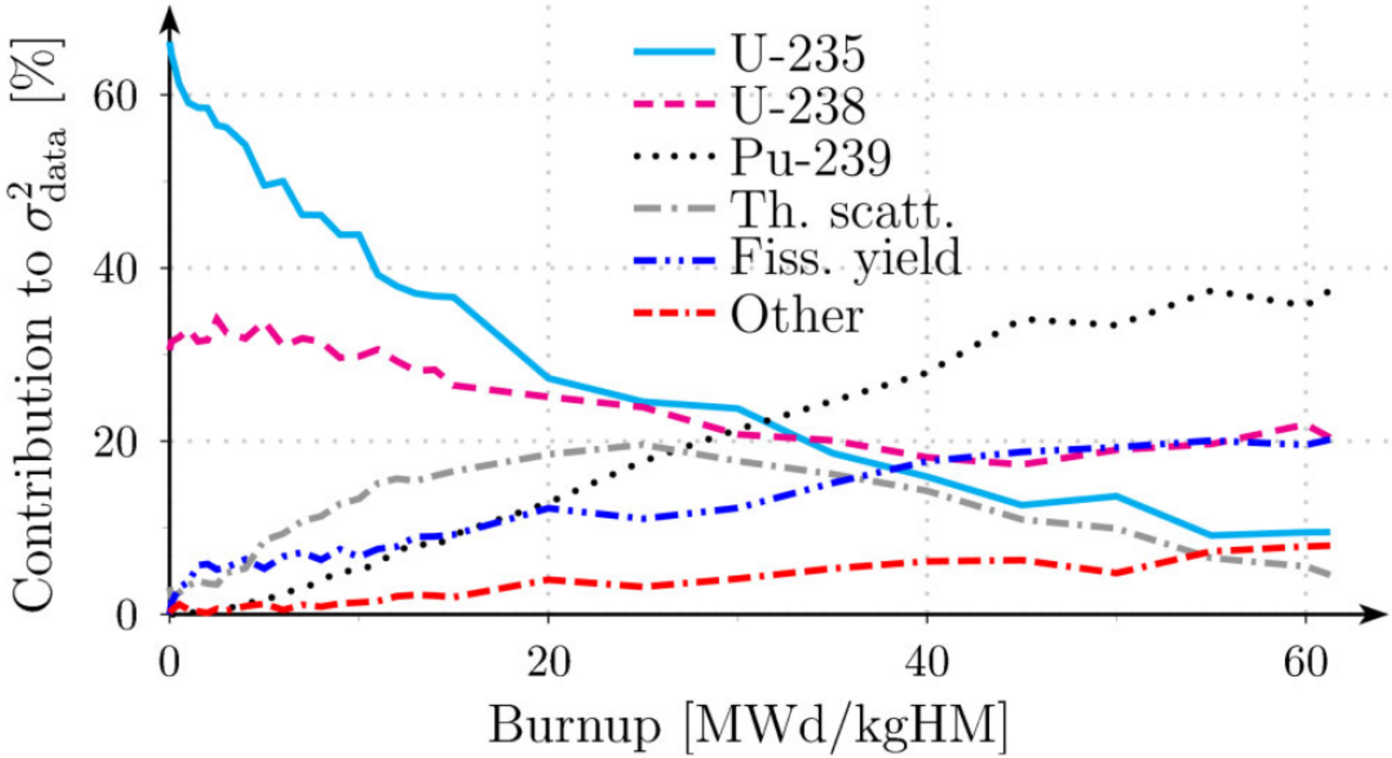
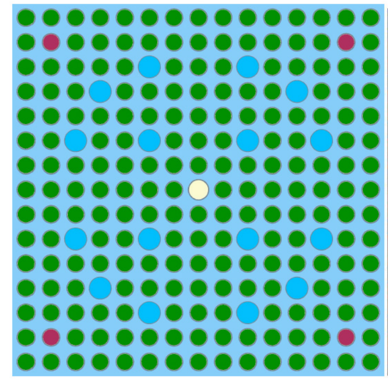


Fig. 4. Contributions to total variance in k_{eff} from variance of individually varied data, for UO_2. "Other" stands for transport and activation data of fission products and minor actinides.

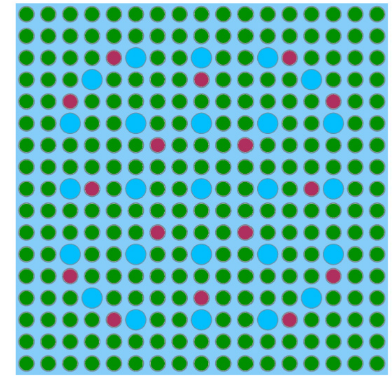
Results 3. Assembly

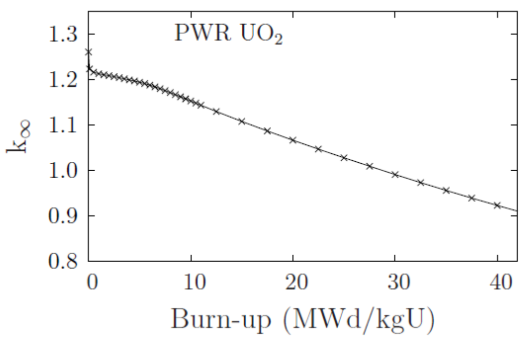
- Different types of assemblies exist: e.g. PWR, BWR, with UO_2, MOX
- It leads to different uncertainties
- Only 2D models are usually used

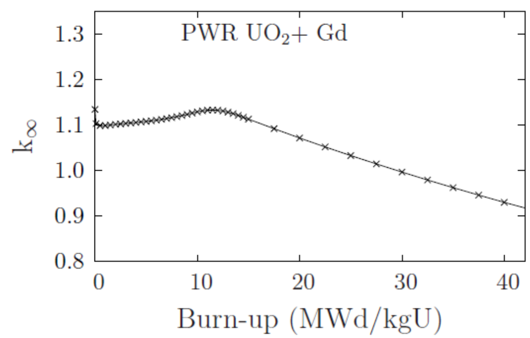
PWR UO_2



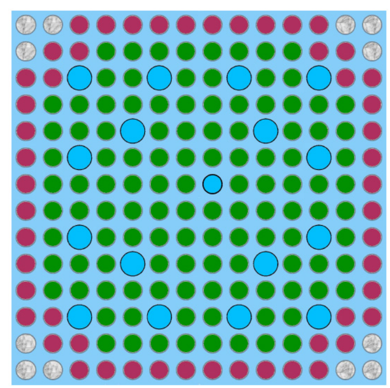
PWR UO_2 + Gd



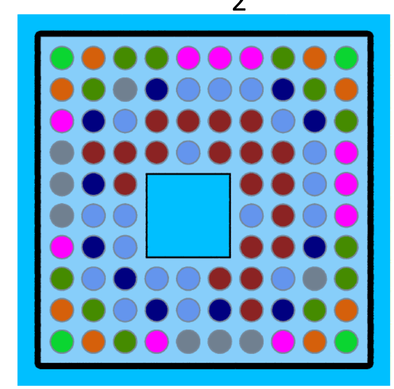


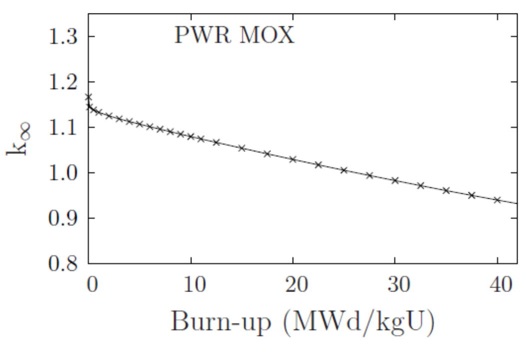


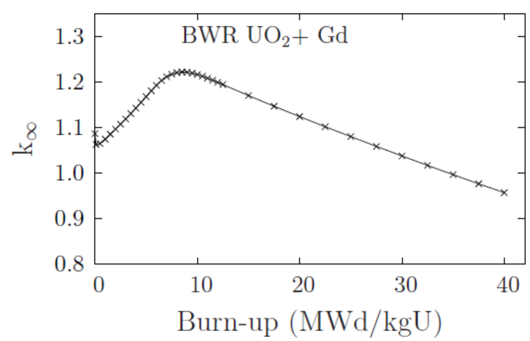
PWR MOX



BWR UO_2+ Gd

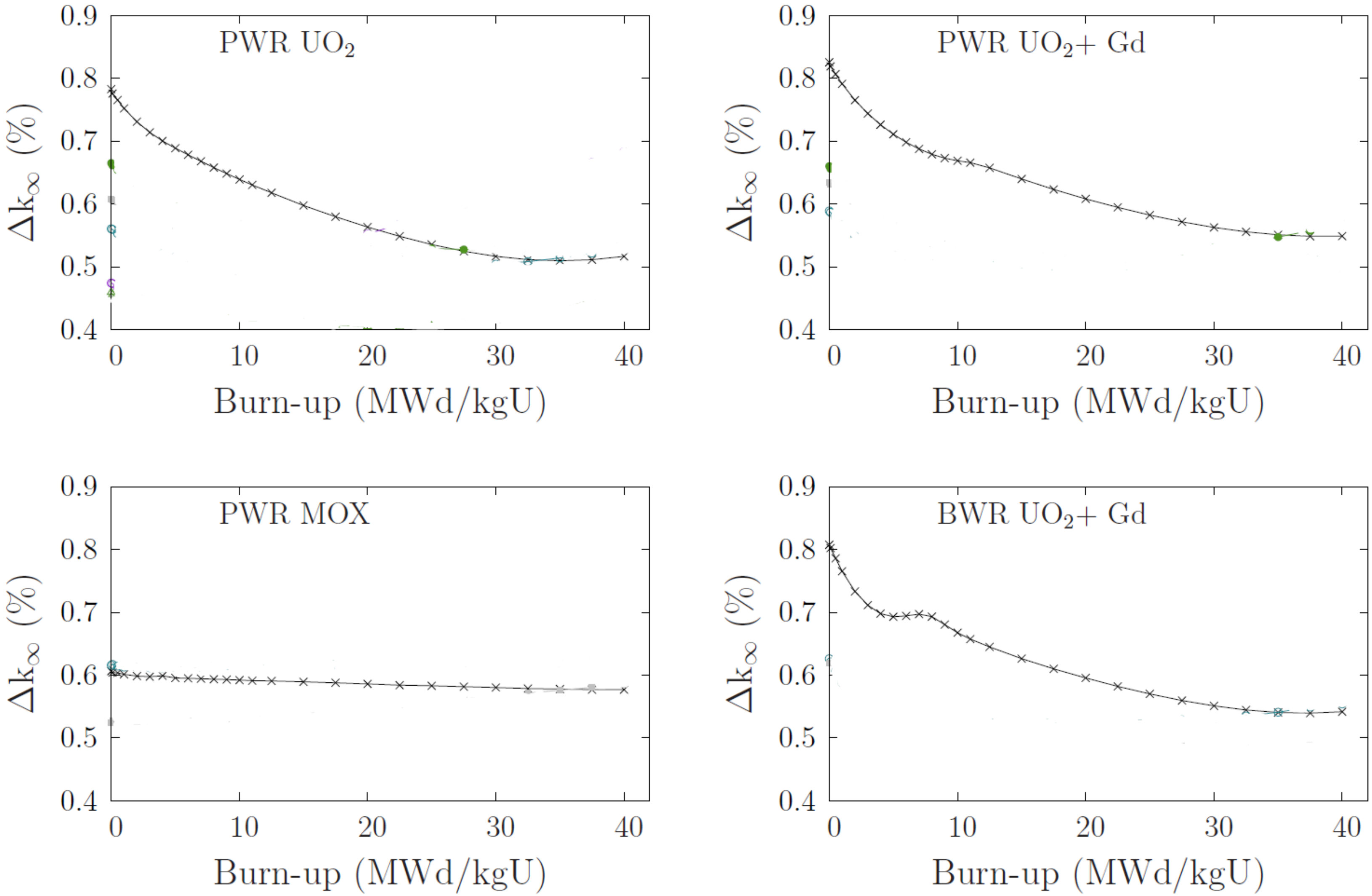






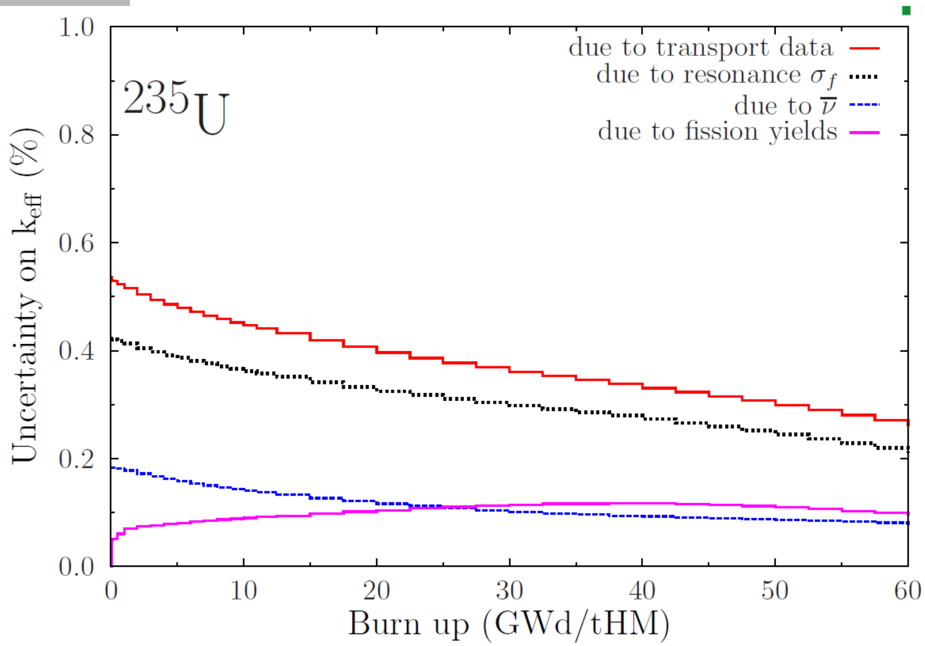
Results 3. Assembly

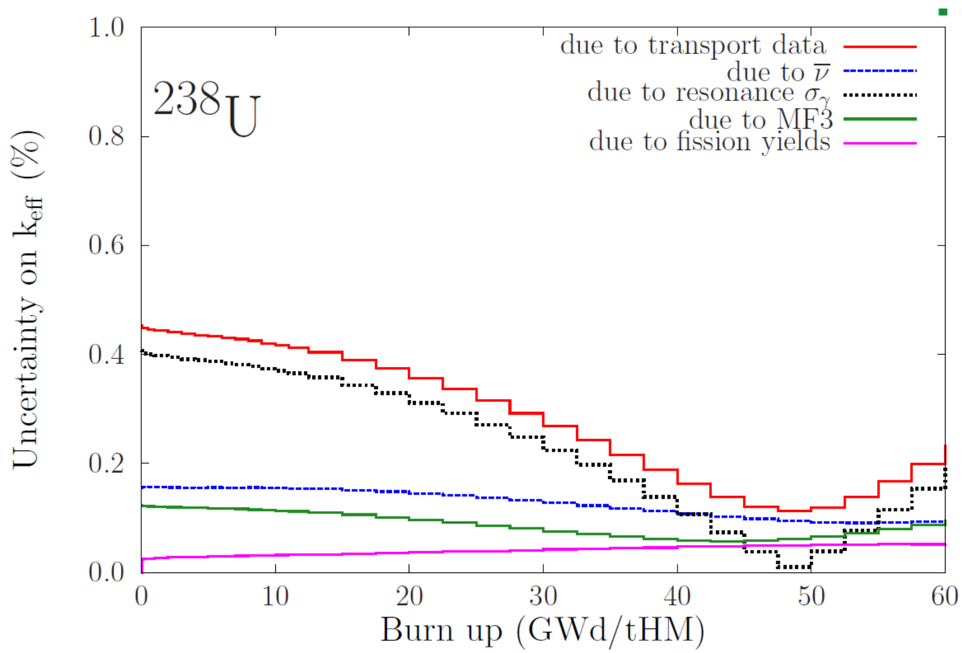
- K_{inf} uncertainty for 4 assemblies, 1 reactor cycle



Results 3. Assembly

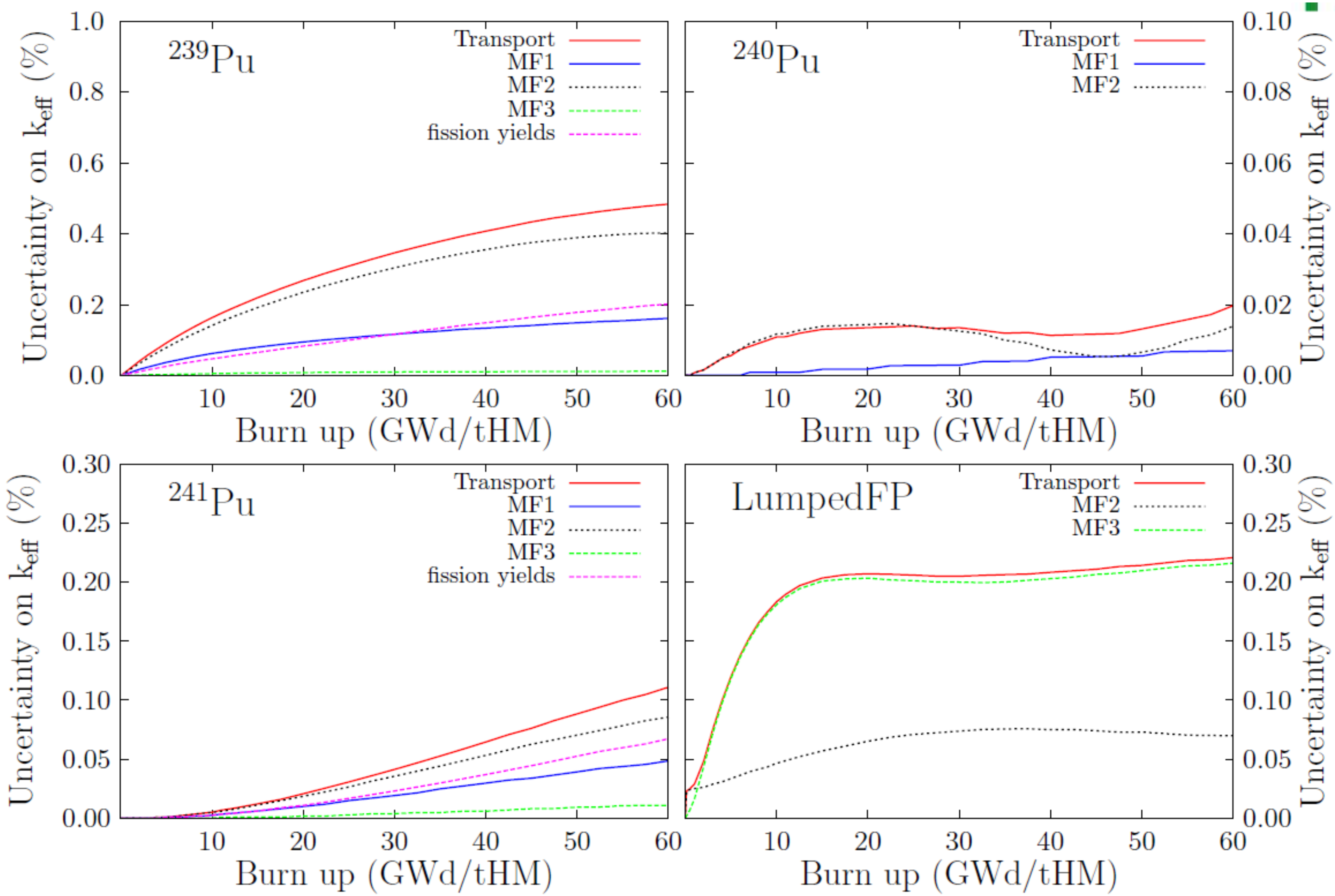
- K_{inf} uncertainty contributions





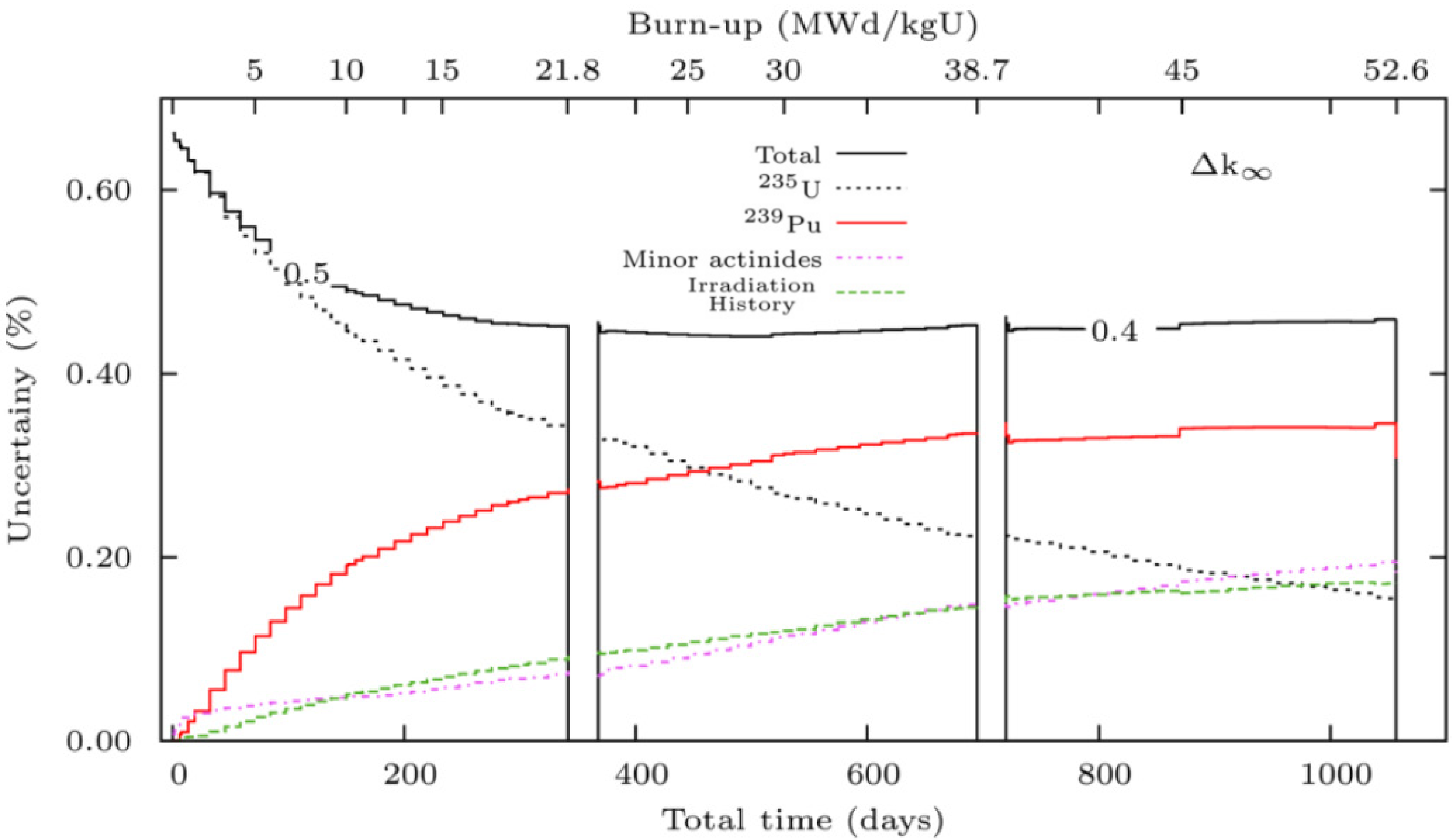
Results 3. Assembly

- K_{inf} uncertainty contributions



Results 3. Assembly

- K_{inf} uncertainty for a PWR UO_2, over 3 successive reactor cycles



Results 4. Full core

- Uncertainty at different locations due to nuclear data,
- For different quantities (inventory, boron concentration, power distribution, safety parameters such as Linear Heat Generation rate…)
- No sensitivity method can be applied

CYCLE MOX

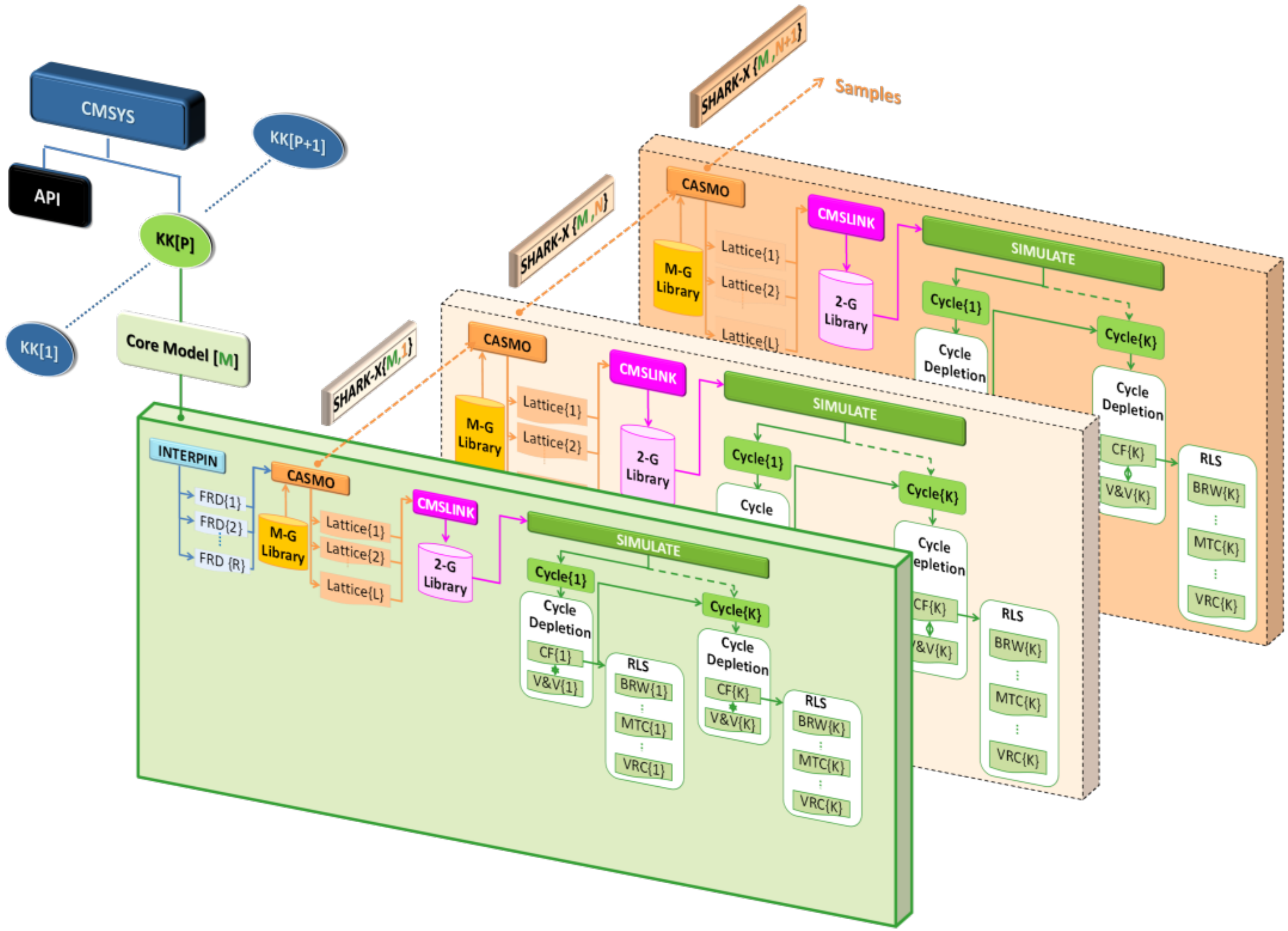
	1	2	3	4	5	6	7	8	9	10	11	12	13
A						29	34	30					
B				30	0	0 *	10	0 *	0	27			
C			30	0	8 *	32	6	32	8 *	0	30		
D		27	0	15	19	9	30	9	19	15	0	30	
E		0	8 *	19	17 *	29	10 *	28	17 *	19	8 *	0	
F	30	0 *	32	9	28	10	17	10	29	9	32	0 *	29
G	34	10	6	30	10 *	17	27 *	17	10 *	30	6	10	34
H	29	0 *	32	9	29	10	17	10	28	9	32	0 *	30
I		0	8 *	19	17 *	28	10 *	29	17 *	19	8 *	0	
J		30	0	15	19	9	30	9	19	15	0	27	
K			30	0	8 *	32	6	32	8 *	0	30		
L				27	0	0 *	10	0 *	0	30			
M						30	34	29					
	1	2	3	4	5	6	7	8	9	10	11	12	13

CYCLE UOX

	1	2	3	4	5	6	7	8	9	10	11	12	13	
A						57	55	57						A
B				47	38	0 *	0	0 *	37	47				B
C			55	0	11 *	47	51	47	11 *	0	55			C
D		47	0	37	36	11	38	11	36	37	0	47		D
E		37	11 *	36	25 *	24	25 *	24	25 *	36	11 *	38		E
F	57	0 *	47	11	24	25	39	25	24	11	47	0 *	57	F
G	55	0	51	38	25 *	39	53 *	39	25 *	38	51	0	55	G
H	57	0 *	47	11	24	25	39	25	24	11	47	0 *	57	H
I		38	11 *	36	25 *	24	25 *	24	25 *	36	11 *	37		I
J		47	0	37	36	11	38	11	36	37	0	47		J
K			55	0	11 *	47	51	47	11 *	0	55			K
L				47	37	0 *	0	0 *	38	47				L
M						57	55	57						M
	1	2	3	4	5	6	7	8	9	10	11	12	13	

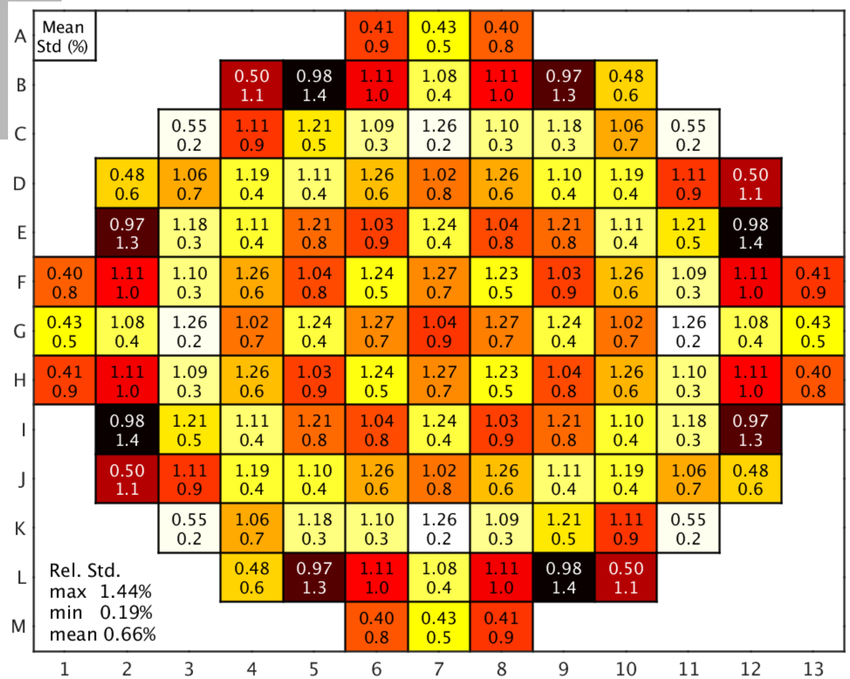
Results 4. Full core

- Example with CASMO/SIMULATE,

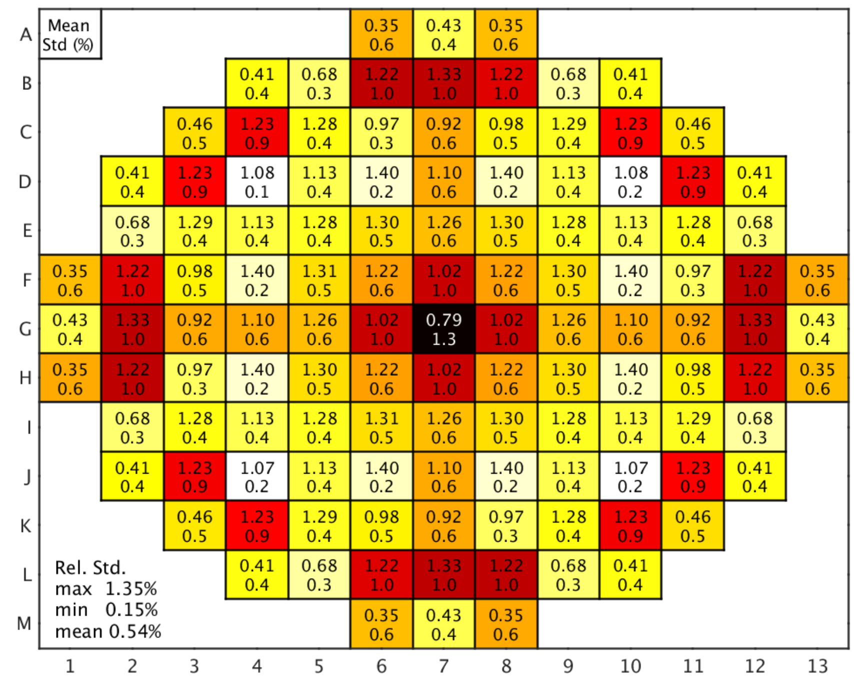


Results 4. Full core

- Example with CASMO/SIMULATE,



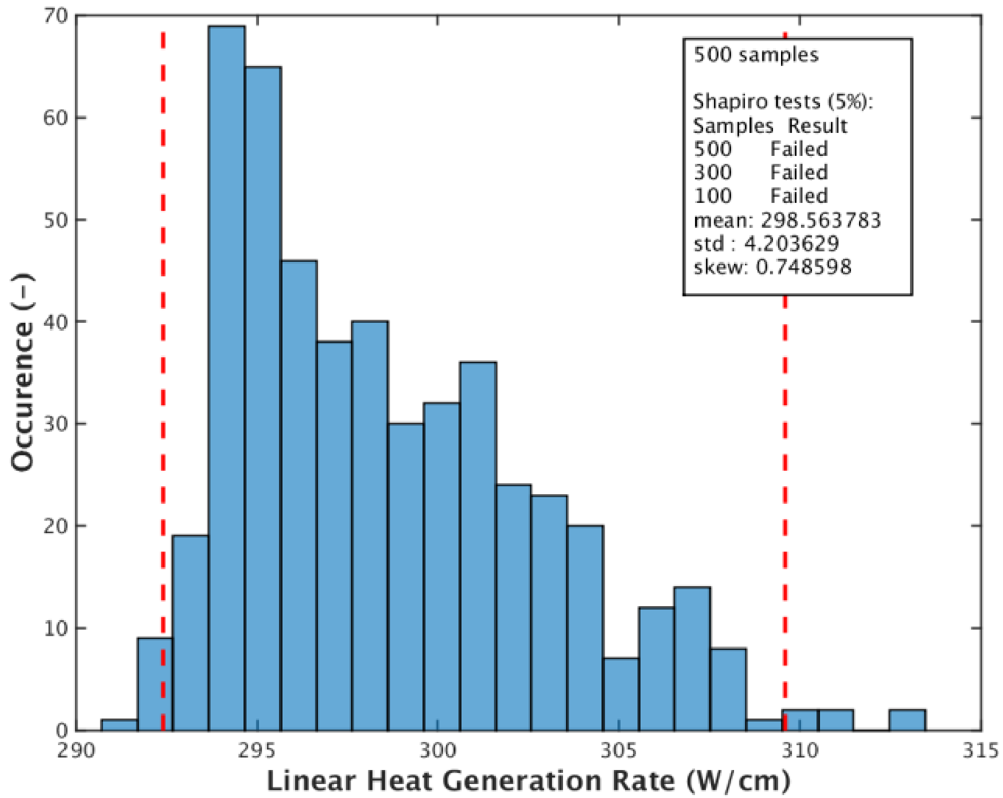
Relative radial power distributions of the MOX



Relative radial power distributions of the UO_2

Results 4. Full core

- Asymmetric distributions for safety parameters



Results 6. Transient

- Control Rod Ejection Accident, with ND uncertainties ($^{235,238}U$, ^{239}Pu, thermal scattering)

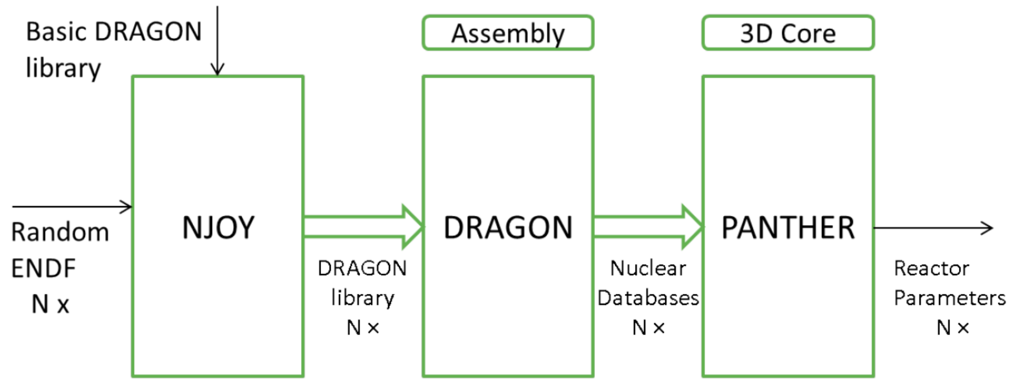


Figure 1. Calculation scheme for the determination of the uncertainties in the main reactor parameters due to nuclear data uncertainties.

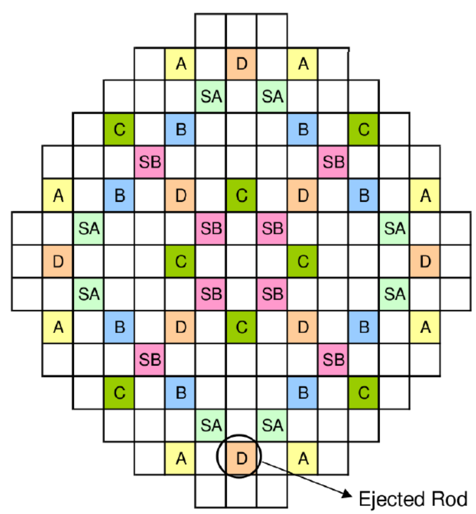
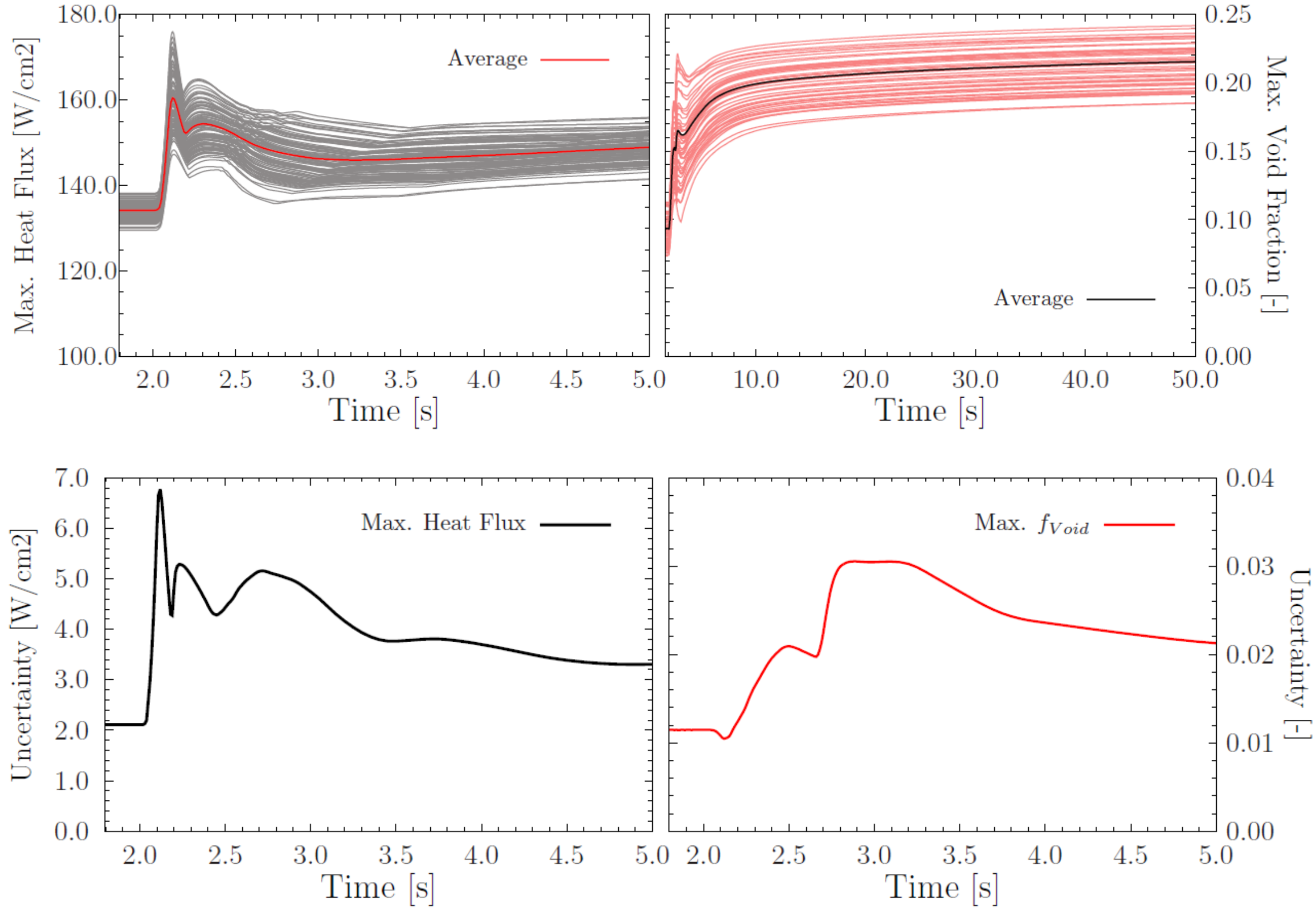


Figure 2. Scheme of Westinghouse core with distribution of control rod banks and position of the ejected control rod.

Results 6. Transient

- Control Rod Ejection Accident, with ND uncertainties



Uncertainty from methods

"Among different participants, given a model definition, which uncertainties do we obtain ?

How are the spread of uncertainties compared to the uncertainties themselves ?"

- Uncertainties due to nuclear data are larger than from many other sources,

1. Sources of nuclear data uncertainties vary: JEFF, ENDF/B, JENDL,TENDL,SCALE, in-house…
2. Processing of nuclear data vary,
3. Methods of uncertainty propagation vary: deterministic, Monte Carlo,
4. Methods of neutron transport/depletion also vary.

- This approach is then different than the UAM requirements,
- It is close to a real-case assignment given by a third party to a TSO (Technical Support Organization).

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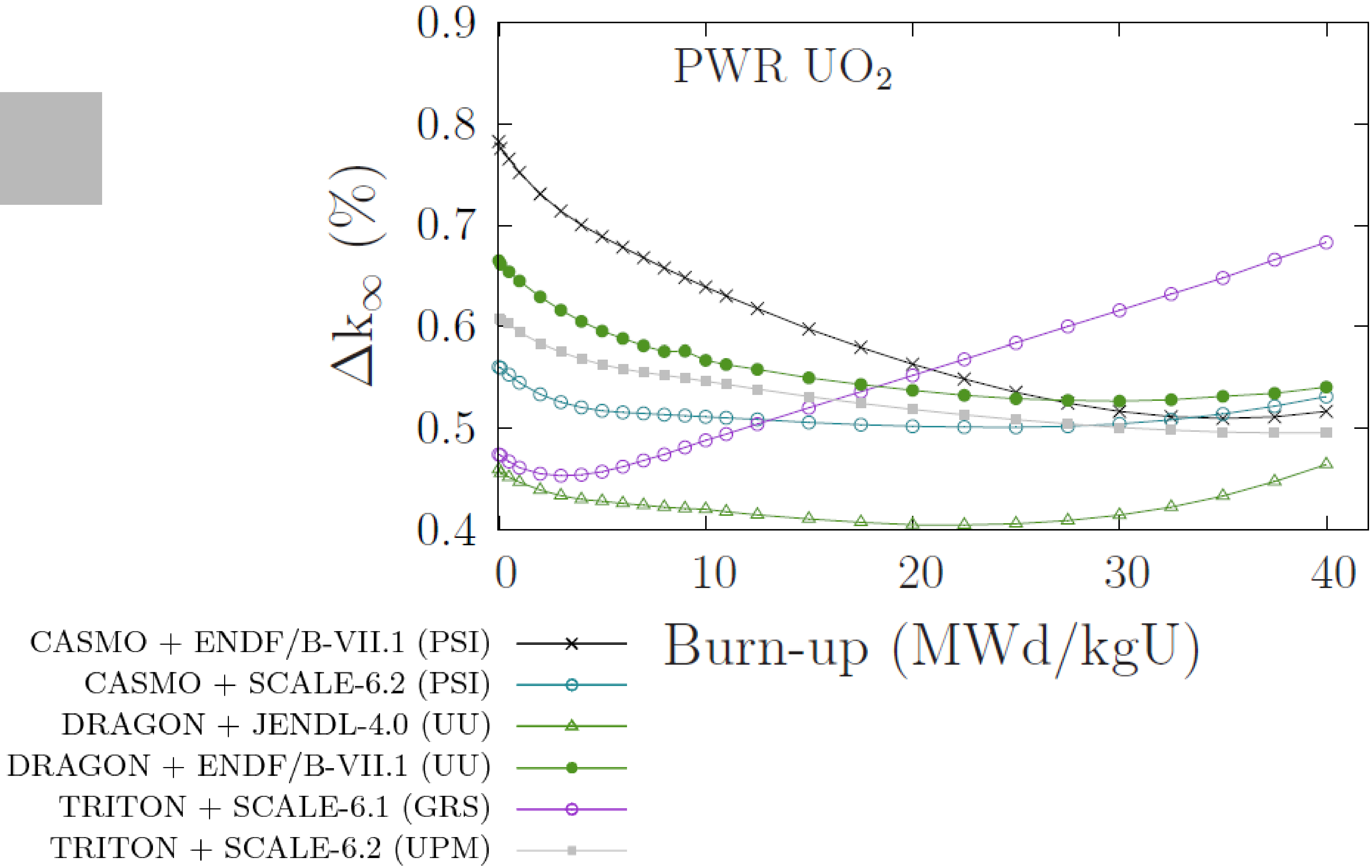
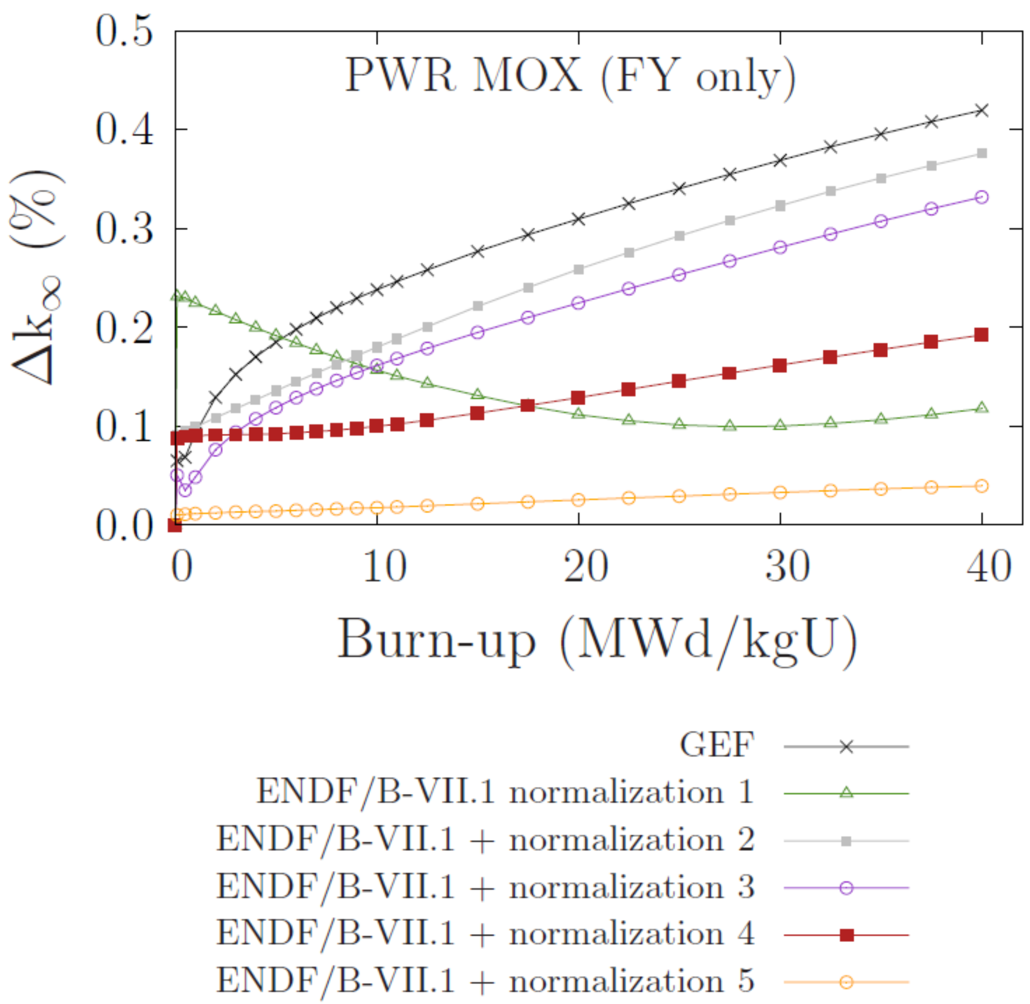
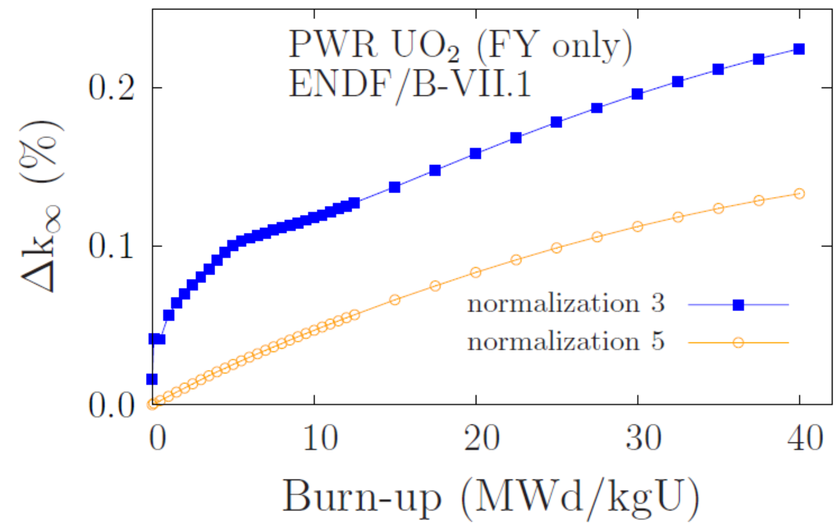
Uncertainty from methods



Uncertainty from methods





Normalization 1:	mass & charge
Normalization 2:	GEF correlation
Normalization 3:	Updated GEF (see [1])
Normalization 4:	∑FY=2
Normalization 5:	SANDY

Other Uncertainties

- For assembly/reactor calculations, other sources of uncertainties appear:
  – Nuclear data,
  – Reactor operating conditions,
  – Manufacturing tolerances,
  – Burnup induced technological changed,
  – …
- All play a role for the assessment on the final quantities

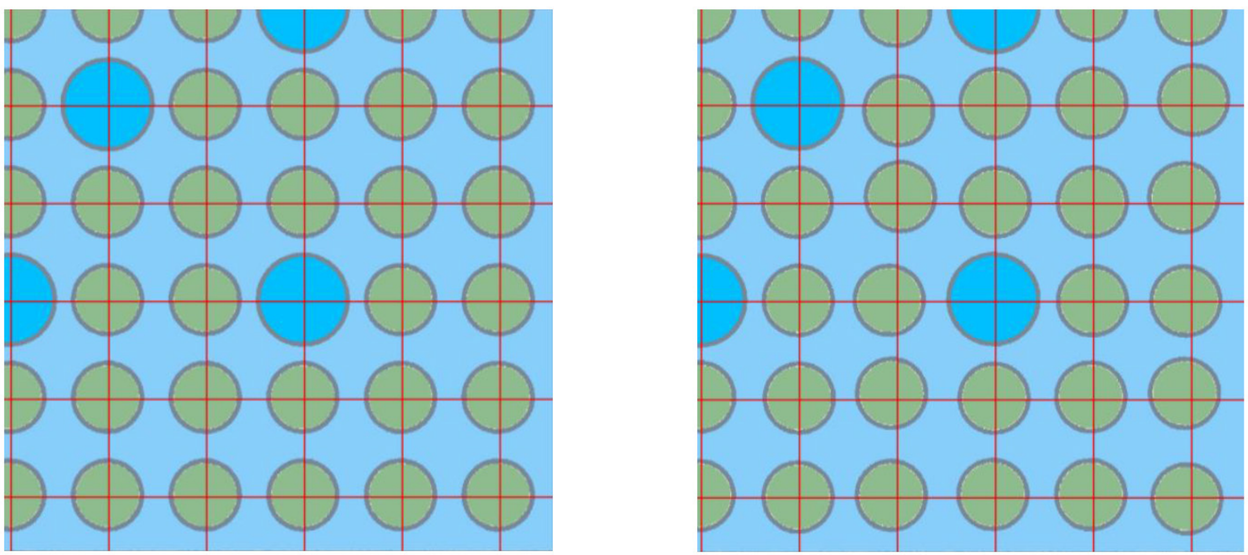
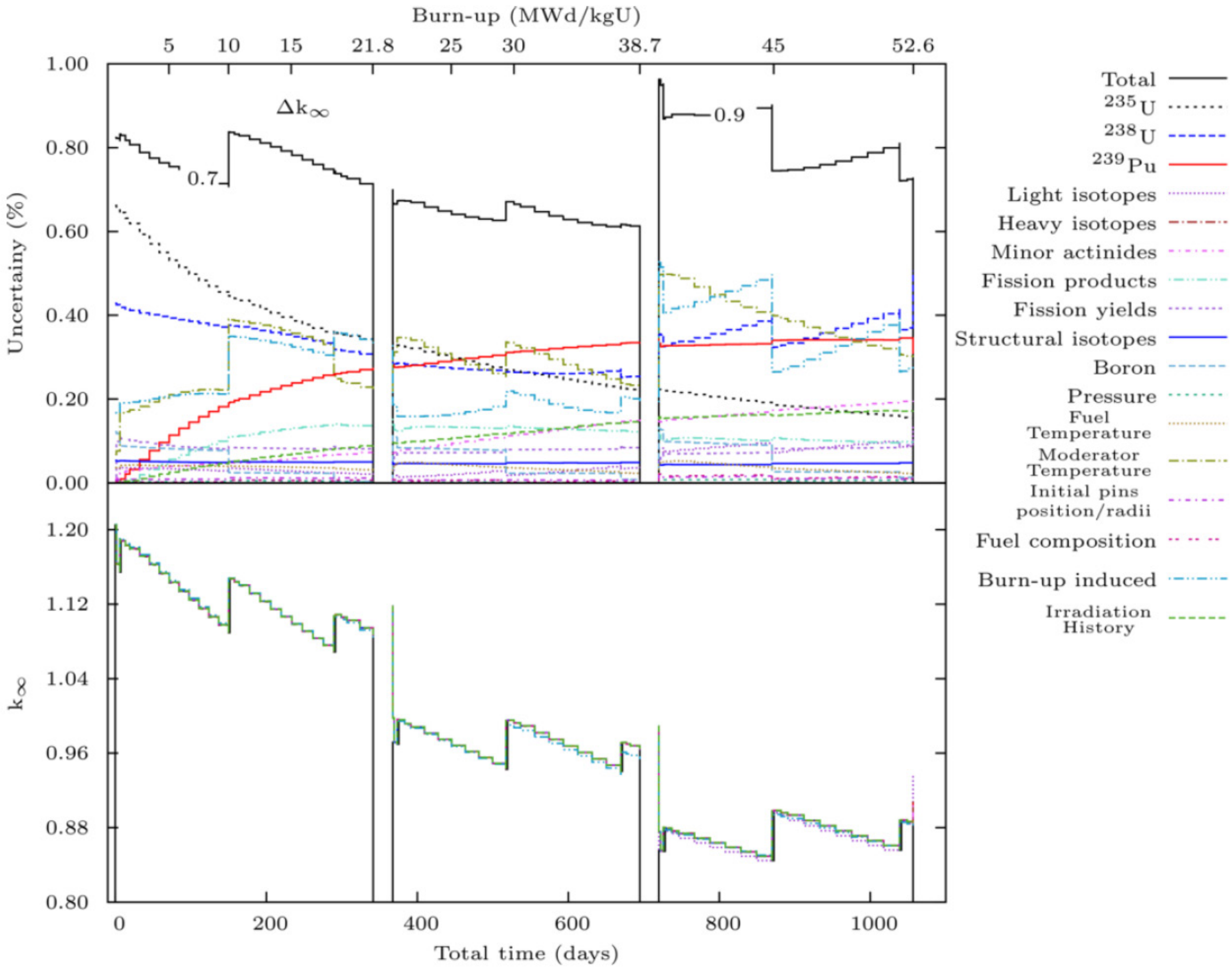
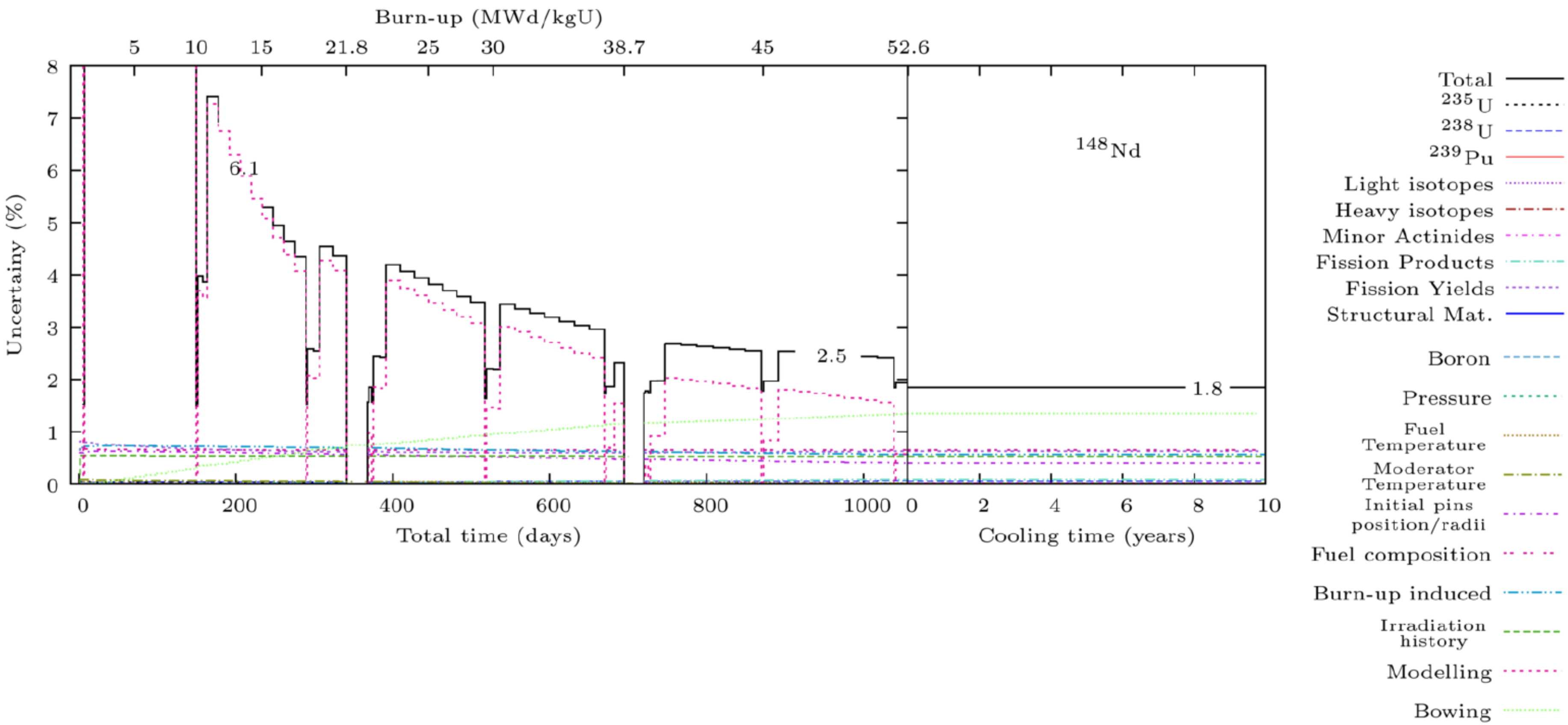


Figure 9: Two random distributions of fuel pins with different enrichments and densities. The colors indicate different fuel pins.

Other Uncertainties

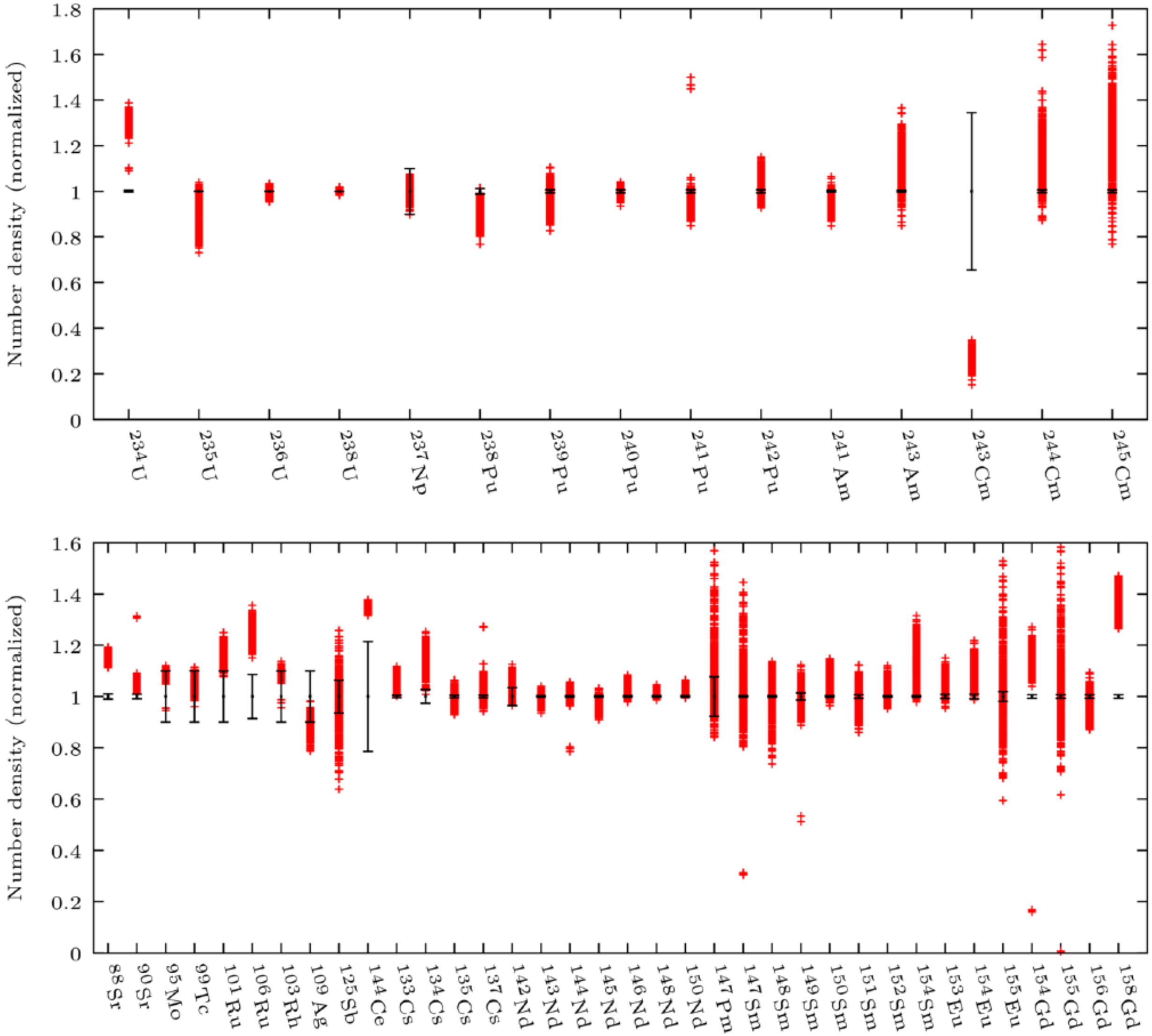


Other Uncertainties



Other Uncertainties

- Finally, the analysis of the total uncertainties help to explain possible differences in C/E ratios



Conclusion

1. Nuclear data uncertainties can nowadays be propagated in large-scale systems, to any quantities

2. A necessary condition is to be able to randomly change the nuclear data (not possible if hardcoded in simulation codes).

3. Other sources of uncertainties exist

4. Finally, uncertainties should be replaced by pdf.

The spread of uncertainties can be higher than the uncertainties themselves (because of methods, sources of data, codes…).
This puts in perspective calculated uncertainties.

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